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KAFEPA: A Computer Code for CANDU PHWR-Fuel Performance Analysis under Reactor Normal Operating Condition

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KAFEPA: 월성로형 핵연료봉의 정상상태 성능분석용 전산코드

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Abstract

A computer code, KAFEPA, for analyzing in-reactor behavior of a PHWR-fuel rod under reactor normal operating condition was developed. This code, KAFEPA, corresponds to the ELESIM code that was developed for the same purpose by AECL. Even though the KAFEPA originated from the ELESIM, it contains more accurate and theoretical models in comparison with the ELESIM, such as fission gas release model, in-reactor densification model and a new database for neutron flux depression across the radial direction in a fuel pellet. The KAFEPA code was verified by comparing the predictions with 22 measurements of fission product gas release. The predictions of the KAFEPA was well agreed with the experimental data.

요 약

월성로형 핵연료봉의 정상상태 노내거동 분석용 전산코드인 KAFEPA를 개발하였다. KAFEPA 전산코드는 같은 목적하에서 AECL에 의해 개발되었던 ELESIM 전산코드에 대응하지만, KAFEPA 전산코드는 ELESIM에 비해 보다 이론적이고 정확하게 예측하는 계산모형들, 즉, 핵분열기체방출 모형, 노내고밀화모형 및 중성자속 감소 계산모형들을 내포하고 있다. KAFEPA 전산코드는 핵분열생성물 기체 방출에 대한 22개 노내 실험자료에 그 예측치를 비교함으로써 검증되었다. KAFEPA 전산코드에 의한 예측치는 상기 실험자료와 잘 일치하였다.

1. Introduction

ELESIM[1] is a representative computer code for in-reactor performance analysis of a CANDU PHWR-fuel rod under reactor normal operating condition. Even though the ELESIM code was adequate for providing information for CANDU PHWR-fuel design, a more precise computer code was required for designing advanced CANDU PHWR-fuel. So, KAERI has developed a computer code, KAFEPA (KAERI/AECL Fuel Element Performance Analysis), based on the ELESIM code, with the aim of not only removing unnecessary conservatism which is imposed on fuel design too much, but of reliably guaranteeing in-reactor fuel soundness. The major difference between the KAFEPA and the

ELESIM is that the KAFEPA code contains KAERI-developed models for fission product gas release and in-reactor densification of UO_2 fuel, and a new database for neutron flux depression across the fuel radius.

In what follows some important information on the KAFEPA code is given in detail, and the predictions of the KAFEPA are compared with in-reactor experimental data and with the predictions of the ELESIM.

2. General Description of the Code

KAFEPA is a fast-running computer code for CANDU PHWR-fuel in-reactor performance analysis under reactor normal operating condition. It computes the one-dimensional temperature distribution, pellet deformation (due to thermal expansion, swelling and densification), gap conductance between UO_2 pellet and Zircaloy clad, fission gas release into free voidage in a fuel rod, rod internal gas pressure, and clad creep deformation. The gas pressure inside the fuel element of a CANDU reactor is usually much lower than the external coolant pressure and the clad is thin (~0.42mm), thus, the clad collapses on contact with the fuel and no fuel-to-clad gap exists. Therefore, pellet relocation due to cracking, and an open gap between the fuel pellet and the clad are ignored in the KAFEPA code. Table 1 shows submodels employed in the KAFEPA code.

Table 1. Submodels Employed in KAFEPA Code

FOR PELLET GEOMETRY
—100 concentric annuli with equal thickness
—half-pellet model to account for the effect of dish on axially thermal expansion; no axial deformation at pellet midplane
FOR THERMAL PERFORMANCE
—1-D Laplace equation in FDM approximation
—Campbell's gap conductance model[2]
—MATPRO model[3] for thermal conductivities of UO_2 pellet and Zry-4
—database for neutron flux depression across UO_2 fuel obtained by KAERI[4]
FOR PELLET DEFORMATION
—two zone model[5] to account for the thermally plastic and non-plastic deformation of UO_2 fuel
—KAERI-developed mechanistic model[6] for in-reactor densification of UO_2 fuel
—the swelling model[7] developed by KAERI
FOR FISSION PRODUCT GAS RELEASE
—KAERI-developed comprehension model[7] for fission product gas release
FOR CLAD CREEP DEFORMATION
—*Hosbon's simulation model*[8]

3. Descriptrion of Selected Submodels

3.1. Pellet Densification

Densification is defined as in-reactor volume change of the UO_2 fuel due to pore shrinkage and/or disappearance without external stresses. As presented in Table 1, the KAERI-developed mechanistic model[6] is employed in the KAFEPA code for the prediction of in-reactor densification of UO_2 fuel, which was developed not only by considering vacancy generation and migration in grains but also by considering the trapping of the vacancies by coarse pores and the growing of the coarse pores. In-reactor densification is function of irradiation time, fission rate, temperature, density, pore size distribution and grain size, as explained in reference 6. Since the initial pore size distribution is the most important parameter for determining the densification rate, the initial intragranular porosity is divided into two groups such as coarse pore and fine pore in the KAFEPA code. And the coarse pore is further characterized by five size ranges.

Fig. 1 illustrates that the mechanistic model predicts fractional fuel volume change in correspondence with experimental data, and it shows that the ELESIM densification algorithm predicts the fraction which is monotonically rise with increasing temperature, while the mechanistic

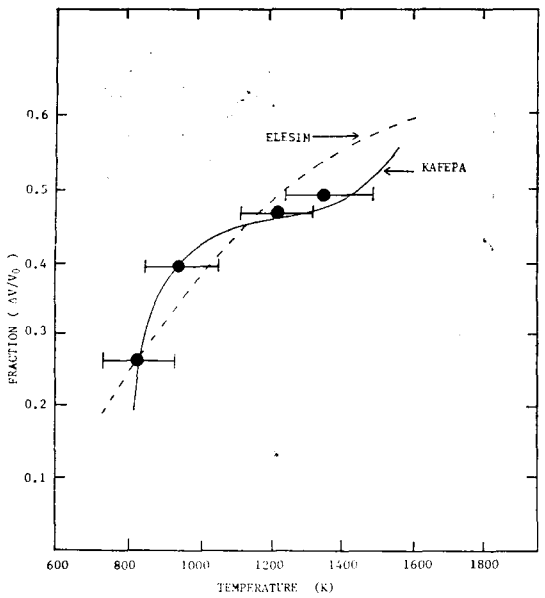

Fig. 1. Fraction ($\Delta V/V_0$) As Function of Temperature for Pickering Element No. 09794 Irradiated to 200±15 MWh/kg-U

model predicts the fraction which is trended as the experimental behaviors.

3.2. Fission Product Gas Release and Fuel Swelling

A KAERI-developed comprehension model[7] for fission gas release is employed in the KAFEPA code, as presented in Table 1. Gas atom diffusion and bubble migration within grain boundary, intergranular bubble formation and size distribution, and intergranular bubble interlinkage were considered in this model as a sequence of events leading to release. These phenomena were applied into the model theoretically by physical bases, i.e., the critical bubble size, the condition for bubble interlinkage and the average bubble size for each size group on the grain boundary which are varied according to the reactor operating condition. To produce the condition of bubble interlinkage, the fraction of total cross-sectional area for lenticular-shaped bubbles on grain boundary to surface area of grain boundary is $\pi/4$. Intergranular bubbles are supposed to be divided into several groups by means of a function for intergranular bubble size distribution, on the basis of their atomic size from diatom to critical bubble size under given circumstances. In performing this, the following assumptions are made: bubble coalescences occur by bubble collisions due to their random migration and intergranular bubbles exceeded the critical bubble size at given circumstances are migrates toward the fuel center or the free voidage along the grain boundary by a driving force due to a thermal gradient.

Fig. 2 shows a calculational scheme for fission gas release model in the KAFEPA code. It is generally accepted that the gaseous fission products are released by mechanisms of recoil, knockout and thermal diffusion. Recoil and knockout are important processes for releasing fission products at low temperatures. These mechanisms are also considered in the KAFEPA code. As shown in Fig. 2, fission gas bubbles on the grain boundary are supposed to be released through a bubble interlinkage tunnel into free voidage in a rod if the average radius of the intergranular bubbles in each size group exceeds critical bubble size at the given circumstance. Swelling due to gaseous fission products on the grain boundary is calculated by considering the bubble saturation condition.

3.3. Neutron Flux Depression

The variation with burnup of the radial power profile in a fuel pellet is a complicated function of both pellet design parameters—fuel geometry and initial enrichment—and reactor operating conditions. As irradiation proceeds, the original fission atoms are depleted, fission products build up and new fissile atoms are formed. In practice, the buildup of plutonium atoms near the fuel surface has the greatest effect on the CANDU-PHWR fuel which is used natural uranium, producing higher fission rates at the surface and reducing fuel center temperature as burnup

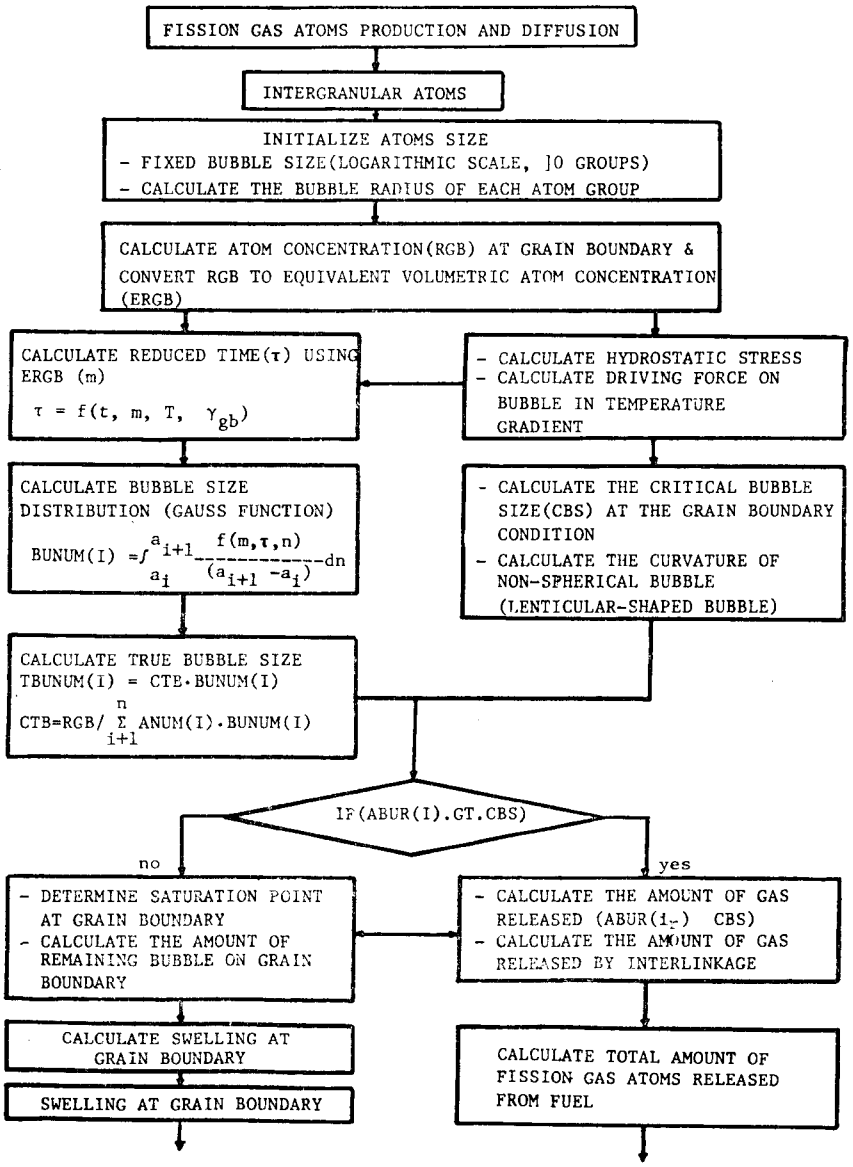

Fig. 2. Calculational Scheme for Fission Products Released by Thermal Diffusion

proceeds.

The HAMMER physics code was run[4] with the help of AECL for a variety of pellet diameters (8.0~19.5mm) and initial fuel enrichments (0.71~6.0wt.% U-235) to obtain tables for the fission distribution as a function of radial position in the fuel at a long range of burnup (0~840 MWh/kg-U). These data are stored in the KAFEPA. An interpolation routine then selects the appropriate set of data at each burnup interval for the paticular fuel pellet diameter and initial fuel enrichment under consideration, and the temperature distribution is calculated using the actual fission distribution.

Fig. 3 shows a comparison of radial power profiles in a UO_2 fuel calculated by the KAFEPA and the ELESIM, respectively, As shown in Fig. 3, the maximum normalized power calculated by

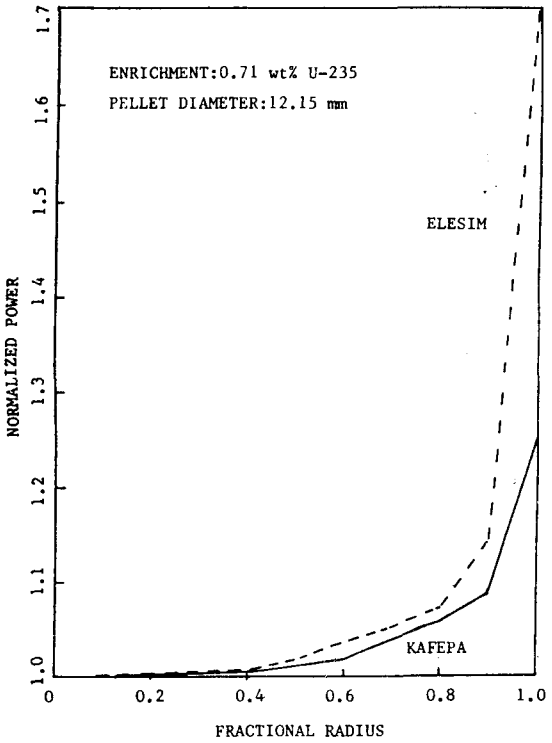

Fig. 3. Comparison of the Radial Power Profiles calculated by ELESIM and KAFEPA Codes: Burnup 120 MWh/kg-U, Enrichment 0.71wt.% U-235

the KAFEPA is 1.25 at 120MWh/kg-U, while that calculated by the ELESIM is 1.7. It is interpreted that the high value, 1.7, calculated by the ELESIM was overpredicted.

4. Verification of the Code

The KAFEPA computer code was verified by comparing the predictions with 22 measurements of fission product gas release obtained from AECL using CANDU PHWR-fuel, including KAERI-manufactured fuel. Fig. 4 shows a comparison of predictions of the KAFEPA code with the experimental data and with those of the ELESIM code.

As shown in Fig. 4, the prediction of the KAFEPA code was in good agreement with the experimental data, while the ELESIM code

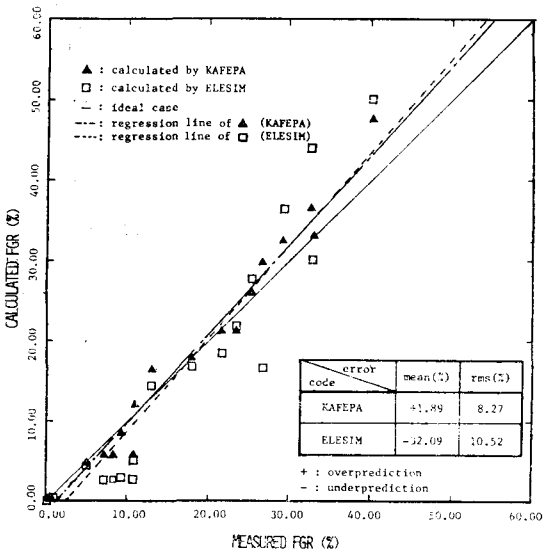

Fig. 4. An Evaluation of the KAFEPA Computer Code

strongly underpredicted in comparison with the experimental data; mean and rms errors in the prediction of the KAFEPA code were 1.89% (overprediction) and 8.27%, respectively, while those in the prediction of the ELESIM code were 32.09% (underprediction) and 10.52%, respectively. From this result, it is certaintly realized that the KAFEPA is much more accurate than the ELESIM.

5. Conclusion

A computer code, KAFEPA, for analyzing in-reactor bethavior of a PHWR-fuel rod during reactor normal operating was developed. This computer code, KAFEPA, corresponds to the ELESIM computer code that was developed for the same purpose by AECL. Even though the KAFEPA, originated from the ELESIM, it contains more accurate and theoretical models in comparison with the ELESIM, such as fission gas release model, in-reactor densification model and a new database for neutron flux depression across the pellet radius.

KAFEPA was verified by comparing the predictions with 22 measurements of fission product gas release. The results indicated that the predictions of KAFEPA were in better agreement with the experimental data than those of ELESIM; mean error and rms error in the prediction of the KAFEPA code were 1.89% (overprediction) and 8.27%, respectively, while those in the prediction of the ELESIM code were 32.09% (underprediction) and 10.52%, respectively. From this, it was found that the KAFEPA is a powerful and accurate computer code for simulating the behavior of UO_2 fuel elements under normal operating conditions of a CANDU-PHWR.

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